NON-PUBLIC?: N
ACCESSION #: 9304120009
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Quad Cities Unit Two PAGE: 1 OF 07

DOCKET NUMBER: 05000265

TITLE: Unit Two Manual Reactor Scram Due To 3E Relief Valve
Failure To Close
EVENT DATE: 03/06/93 LER #: 93-006-00 REPORT DATE: 04/01/93

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 4 POWER LEVEL: 071

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR
SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: Mark L. Huhn, Ext. 2117 TELEPHONE: (309) 654-2241

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SB COMPONENT: RV MANUFACTURER: D245
REPORTABLE NPRDS: Y

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On March 6, 1993, Unit Two was in the RUN mode at 71% of rated core thermal power. At 2019 hours, the 2-203-3E Electromatic Relief Valve (ERV) RV! failed to close during routine testing. At 2021 hours, the reactor RCT! was manually scrammed per procedure and the relief valve reseated on its own at 193 psig reactor pressure. The Unit was in cold SHUTDOWN at 0112 hours on March 7, 1993. During a Drywell inspection it was discovered that the leakoff line from the 2-203-3E ERV was sheared off where it attaches to the 8 inch discharge header. NRC notification was completed at 2156 hours on March 6, 1993 per 10 CFR 50.72(b)(2)(ii).

Investigation revealed that the cause of the event was component failure. The pilot valve seat bushing had severe steam cutting. The fractured leakoff line piping is being sent off site for failure analysis. This ERV will be replaced with an overhauled valve during the current refuel outage and it will be tested along with all of the other relief valves

during reactor startup.

This report is being submitted in accordance with 10 CFR 50.73(a)(2)(iv).

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END OF ABSTRACT

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PLANT AND SYSTEM IDENTIFICATION:

General Electric - Boiling Water Reactor - 2511 MWt rated core thermal power.

EVENT IDENTIFICATION: Unit Two manual reactor scram due to 3E relief valve failure to close.

A. CONDITIONS PRIOR TO EVENT:

Unit: Two Event Date: March 6, 1993 Event Time: 2021
Reactor Mode: 4 Mode Name: RUN Power Level: 71%

This report was initiated by Deviation Report D-4-2-93-018.

RUN Mode (4) - In this position the reactor system pressure is at or above 825 psig, and the reactor protection system is energized, with APRM protection and RBM interlocks in-service (excluding the 15% high flux scram).

B. DESCRIPTION OF EVENT:

At 2019 hours on March 6, 1993, Unit Two was in the RUN mode at 71% of rated core thermal power. The Operations department was performing QCOS 203-3, "Main Steam Relief Valve Operability Test". After testing the 2-203-3E Electromatic Relief Valve (ERV) RV!, the valve failed to reseat when the control switch was taken to close. Closed light indication was received, indicating that the solenoid SOL! actuator had deenergized. At 2021 hours, the reactor RCT! was manually scrammed per QCOA 203-1, "Failure of a Relief Valve to Close or Reseat Properly", after numerous attempts to reclose the valve.

Reactor water level dropped below +8 inches causing a Group II, III JM!, Reactor Building Vent VA! isolation, Control Room Vent VI! isolation, and an auto start of the Standby Gas Treatment System

BH!.

At 2136 hours, on March 6, 1993, the valve reseated with reactor pressure at 193 psi. The Unit was in cold SHUTDOWN at 0112 hours on March 7, 1993. During the one hour and 15 minutes that the relief valve was open, the reactor cooled down approximately 159 degrees F.

At 2156 hours on March 6, 1993, a four hour Emergency System Notification was made per 10 CFR 50.72(b)(2)(ii).

A drywell entry was made, and it was discovered that the 2-203-3E

ERV leakoff line was sheared off where it attaches to the 8 inch discharge header.

On March 7, 1993, the Mechanical Maintenance (MM) department, under Work Request Q98939, removed the valve from the Unit Two reactor and returned it to the maintenance shop for inspection and overhaul.

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Following inspection and disassembly of the relief valve and pilot valve, it was determined that the valve failed to reseat due to severe steam cutting on the pilot valve seating surface.

C. APPARENT CAUSE OF EVENT:

This report is being submitted in accordance with 10 CFR 50.73(a)(2)(iv). The licensee shall report any event or condition that resulted in a manual or automatic actuation of any Engineered Safety Feature (ESF) JE!, including the Reactor Protection System (RPS) JC!.

The cause of the relief valve failing to close was determined to be severe steam cutting of the ERV pilot valve seat bushing.

In order to determine the cause of the relief valve's failure to close, an extensive inspection was conducted. Following a general visual inspection of the valve prior to its removal from the drywell, the valve was taken to the Maintenance shop for further inspection and disassembly. Initially, water was poured into the Main Disc seating area to determine if the main disc was fully closed. The water did not pass by the main disc seating area indicating that the main disc was in the closed position.

The valve was then mounted in a test stand to perform an "As Found" nitrogen leak test. As pressure was applied to the valve, gross leakage was identified from the pilot valve discharge port. The pilot assembly was removed from the valve and a test pilot was installed to allow further testing of the main seat. As soon as pressure was applied to the valve, the main seat leaked extensively.

The main disc spring was checked by pressing down on the main disc and moving it off of its seat. Spring tension was considered adequate and a strongback was used to hold the main disc open to perform an inspection of the disc and seating surfaces. While there was an accumulation of small particulate debris on the disc and seat sealing surfaces, there was no evidence of steam cutting. It is believed that this debris accumulated during the time the valve was open. This debris would explain the excessive leakage seen during the "As Found" pressure testing. It is not believed that this accumulation of debris on the main disc seat surfaces contributed to the valve sticking open.

The pilot assembly was installed into another test fixture and leak tested. The leakage was so severe that the assembly would not hold pressure. The pilot stem was stroked repeatedly under pressure to simulate an actuation, but at no point would it reseat.

Following the leak test, the pilot assembly was disassembled and inspected. Severe steam cutting on the pilot seat bushing was evident. The pilot disc did not appear to have any significant steam damage.

Steam damage was also noticed on the seat bushing flexitallic gasket. While the damaged gasket would have contributed to pilot valve leakage, it is not believed that this amount of leakage could have caused the relief valve to stick open. In order to determine the reason for the steam damage evident on the gasket, the gasket was removed from the pilot assembly and its thickness was measured.

The average thickness of the gasket was found to be .114". This flexitallic gasket is a .125" thick gasket that is designed to crush to .100", and once removed would flex back to approximately .105".

This suggests that the solenoid bracket stud nuts were not torqued adequately to achieve the recommended crush on the gasket. Testing was performed with a test pilot and a new gasket to determine the torque necessary to crush the gasket to .105". Indications are that approximately 110 ft-lbs of torque were required to achieve this amount of crush. The installation procedure per the valve manufacturer recommends an upper torque limit of 80 ft-lbs. The valve manufacturer has been notified and an engineering study is being conducted to determine the amount of torque which is required to get a proper compression of the gasket.

An inspection was also conducted on the main valve components during disassembly. No major degradation was identified which would have contributed to the valve sticking open.

The operation of the pilot valve is crucial to the overall operation of the ERV. If the pilot fails to adequately seat following an actuation, the main disc will not be able to close.

According to the valve manufacturer, the main disc spring would not be capable of closing the valve by itself until approximately 50 psig reactor pressure. It is believed that as reactor pressure was decreasing, pilot discharge volume was also decreasing, and eventually, reached a point at approximately 190 psig where the main disc was able to build up enough pressure to close the valve with assistance from the main disc spring.

There are several contributing causes to the severe steam cutting of the pilot valve seat bushing. Inadequate handling of the pilot valve assembly prior to its installation into the valve may have caused a misalignment within the pilot valve, which could have led to the seat leakage. Also, the pilot valve testing method may have been insufficient, allowing the valve to have been installed in a somewhat degraded condition.

The pilot valve associated with this event was rebuilt to a revision of the overhaul procedure which did not address the contributing causes discussed above. The root cause of the pilot valve leakage is an inadequate procedure associated with the pilot valve rebuild.

The 2-203-3E relief valve and pilot valve had previously been replaced and successfully tested in April of 1992. The valve was also successfully tested on November 7, 1992. Another potential contributing factor could have been an accumulation of foreign debris on the pilot valve seating surface from when the valve was operability tested during reactor startup.

Another contributing cause to the failure of this relief valve to reseat, may have been the severe vibrations seen at the 3E relief valve position. This valve position has historically been a high vibration area.

D. SAFETY ANALYSIS OF EVENT:

The safety consequences of this event were minimal. All operator actions were performed per approved station procedures to bring the reactor to a safe shutdown condition. The relief valve closed at approximately 190 psig reactor pressure, and the Unit was in cold SHUTDOWN at 0112 hours on March 7, 1993.

The High Pressure Coolant Injection (HPCI) BJ!, Reactor Core Isolation Cooling (RCIC) BN!, Residual Heat Removal (RHR) BO!, and Core Spray BM! Systems were available at all times during the occurrence to supply water to the reactor vessel. However, normal reactor feedwater was adequate to control level.

Immediately after the scram, reactor pressure started to decrease. The coolant temperature dropped 159 degrees F in the one hour and 15 minutes that the relief valve was open. Technical Specification (T.S.) section 3.6.A.1 limits cooldown to 100 degrees F per hour except as specified in T.S. section 3.6.A.2. which allows for a step reduction for 240 degrees F as long as T.S. section 3.6.A.3 is met. T.S. section 3.6.A.3 requires the shell flange temperature to be within 140 degrees F of the shell temperature. The 100 degree F per hour cooldown was exceeded, but the 240 degree F step change and the 140 degree F differential temperature limit were not exceeded. The thermal design cycle analysis for the reactor pressure vessel addresses relief valve blowdowns. The thermal cycle instructions provided by General Electric describe the single SRV blowdown as follows: "Single SRV fails open, followed by a SCRAM and rapid depressurization and cooldown of the vessel." The effects of the SCRAM and rapid depressurization and cooldown are included in the analysis for the blowdown cycle.

The thermal cycle analysis allows for twelve relief valve blowdowns. This is the first event to be counted against this item on Unit 2.

E. CORRECTIVE ACTIONS:

The immediate corrective action was to SCRAM the reactor per QCOA 203-1, "Failure of a Relief Valve to Close or Reseat Properly". An initial drywell inspection was performed and the valve was removed from the reactor and taken to the MM decon shop. The valve was inspected and tested to determine the cause of its failure to reseat properly. The valve will be replaced with a rebuilt valve during the current refuel outage and all of the relief valves will be operability tested during reactor startup.

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It has been verified that all of the pilot valves which will be installed during Q2R12 were rebuilt under the current revision of the pilot valve rebuild procedure. Any pilot valves which were not rebuilt to the current revision of the procedure will be reworked.

Three of the four pilot valves currently installed on Unit One were verified to have been rebuilt to the current revision of the pilot valve rebuild procedure. The only pilot valve not rebuilt to the current rev is the 1-203-3B pilot valve. However, tailpipe temperatures do not indicate that this pilot valve is currently leaking. A work request will be written to change the pilot valve on the 1-203-3B ERV during an outage of sufficient duration (NTS# 2652009301801).

The leak test to be performed on the rebuilt pilot valve assembles will be modified to require a pressure decay test. The listing will also include pilot valve actuations while it is under pressure to verify that the pilot valve will reseat. This new testing method will be performed on the pilot valves to be installed during th

current outage (NTS# 2652009301802).

Protective sleeves have been fabricated and the associated procedure will be revised to require the sleeves to be placed onto the rebuilt pilot valve assemblies for storage after completing a successful pressure test. The pilot assemblies will remain in the protective sleeves until just prior to their installation in the pilot valve housing. This will minimize the potential for misalignment during handling (NTS# 2652009301803).

A revised torque value for the solenoid bracket stud nuts may be added to the pilot valve installation procedure to ensure the flexitallic gasket receives an adequate crush pending the completion of the valve manufacturers engineering study (NTS# 2652009301804).

The system engineer will continue to trend tailpipe temperatures on a regular basis. Information on relief valves with elevated temperatures will be discussed with the Operations department to determine the potential risk involved with the actuation of these valves. Work Requests will be written for valves which have elevated temperatures and the pilot valves will be changed out at the first short outage of sufficient duration.

Engineering Applied Sciences, Inc. will provide recommendations on design changes necessary to eliminate or lessen the vibrations present at the 3E relief valve position (NTS#2652009301805).

F. PREVIOUS EVENTS:

There have been two previous events involving stuck open ERVs in the past five years. Both failures resulted in a manual shutdown of the reactor. One of these failures was attributed to binding of the pilot valve spool. The other event was due to plugging of the drain orifice.

o DVR 04-02-91-073 (LER 91-12) - Unit Two was in the STARTUP mode at one percent rated core thermal power. At approximately 63 psig, a spurious actuation of ERV 2-203-3C occurred. The cause of the event was attributed to binding between the spring cover of the pilot valve assembly -and the solenoid support bracket, causing the pilot valve stem to remain partially depressed.

o DVR 04-01-89-031 (LER 89-004) - While performing relief valve testing the 1-203-3D valve stuck open. The reactor was manually scrammed per procedure. The cause was determined to be plugging of the drain orifice.

A Nuclear Plant Reliability Data System (NPRDS) search was conducted for Dresser Industries 6 inch relief valve, model 1525-VX. A total of six failures were identified at other stations which currently use this type of relief valve. Two of the six failures were due to pilot valve leakage.

G. COMPONENT FAILURE DATA:

Relief valve 2-203-3E is a 6-inch ERV manufactured by Dresser Industries Inc., Model Number 1525-VX.

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Commonwealth Edison

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RLB-93-059

March 31, 1993

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Reference: Quad Cities Nuclear Power Station
Docket Number 50-265, DPR-30, Unit Two

Enclosed is Licensee Event Report (LER) 93-006, Revision 0 For Quad Cities Nuclear Power Station.

This report is submitted in accordance with the requirements of the Code of Federal Regulations, Title 10, Part 50.73(a)(2)(iv). The licensee shall report any event or condition that resulted in manual or automatic actuation of any Engineered Safety Feature.

Respectfully,

COMMONWEALTH EDISON COMPANY
QUAD CITIES NUCLEAR POWER STATION

R. L. Bax

Station Manager

RLB/TB/plm

Enclosure

cc: J. Schrage
T. Taylor
INPO Records Center
NRC Region III

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